Status of Industry Initiative on Heavy Load Lifts

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Topics

- Industry initiative (Pietrangelo to Dyer 9/14/07)
- Status of initiative
  - Phase 1 commitments
  - Maintenance Rule
  - Wet lifts
  - SFP cranes
  - FSAR update
- Discussion of acceptable load drop analysis
  - Comparison of industry initiative drop analysis with NUREG-0612 criteria
  - Criteria for drop analysis methods
  - Component/material acceptance criteria

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Industry Initiative: Prior to 7/1/08

1. Ensure compliance with NUREG 0612, Phase 1
2. For reactor vessel head lifts:
   a) If you have single failure proof crane (equivalent) OR load drop analysis that bounds planned lifts (weight, height, and medium), ensure procedures reflect safety basis
   b) If not, perform "wet lift," where the bottom of the head is < 15 feet above the refueling cavity water surface
3. Ensure maintenance rule (a)(4) administrative controls include movement of heavy loads

Industry Initiative: Post 7/1/08

1. Ensure compliance with NUREG 0612, Phase 1
2. For reactor head lifts and cask lifts over the spent fuel pool, ensure either a single failure proof crane or a load drop analysis that bounds planned lifts
3. Ensure maintenance rule (a)(4) administrative controls include movement of heavy loads
4. Provide a summary description in next FSAR update
5. If load drop analysis is used, ensure restrictions on weight, height, and medium are reflected in procedures

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Status of Industry Initiative

- Task Group formed with representation from utilities, vendors and architect/engineers
  - Holding weekly conference calls
  - Subgroups assigned to develop guidance documents
- Current Status
  - Phase 1 commitments (shared gap analysis procedures)
  - Maintenance Rule (developed guidance)
  - Wet lifts (discussed considerations and shared procedures)
  - Load drop analysis (draft guidance developed)
  - SFP cranes (Dec 5 meeting)
  - FSAR update (start in Jan)
- May hold a workshop in early 2008
  - Share task force progress
  - Obtain feedback on initiative implementation
  - Share best practices

Discussion of Load Drop Analysis
(beyond design basis considerations)

- Comparison of industry initiative drop analysis with NUGEG-0612 criteria
- Criteria for drop analysis methods
- Component/material acceptance criteria

Comparison of Industry Initiative with NUREG-0612

5.1 Evaluation Criteria	Initiative Analysis
I. Releases of radioactive material that may result from damage to spent fuel based on calculations involving accidental dropping of a postulated heavy load produce doses that are well within 10 CFR Part 100 limits of 300 rem thyroid, 25 rem whole body (analyses should show that doses are equal to or less than 1/4 of Part 100 limits);	Demonstrate that after the reactor vessel head drop, the core remains covered with coolant and sufficient cooling is available.
II. Damage to fuel and fuel storage racks based on calculations involving accidental dropping of a postulated heavy load does not result in a configuration of the fuel such that keff is larger than 0.95;	
III. Damage to the reactor vessel or the spent fuel pool based on calculations of damage following accidental dropping of a postulated heavy load is limited so as not to result in water leakage that could uncover the fuel, (makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost is borated); and	
IV. Damage to equipment in redundant or dual safe shutdown paths, based on calculations assuming the accidental dropping of a postulated heavy load, will be limited so as not to result in loss of required safe shutdown functions.	

Comparison of Industry Initiative with NUREG-0612 Appendix A

Appendix A 1. General Considerations	**Initiative Analysis**
(1)That the load is dropped in an orientation that causes the most severe consequences	The reactor vessel head drop is concentric and impacts directly on the vessel flange.
(2) That fuel impacted is 100 hours subcritical (or whatever the minimum that is allowed in facility technical specifications prior to fuel handling)	N/A
(3) That the load may be dropped at any location in the crane travel area where movement is not restricted by mechanical stops or electrical interlocks	The reactor vessel head is dropped directly above the vessel at the maximum height controlled by plant procedures.
(4) That credit may not be taken for spent fuel pool area charcoal filters; if hatches, wall, or roof sections are removed during the handling of the heavy load being analyzed, or whenever the building negative pressure rises above (-)1/8 inch (-3 m) water gauge	N/A
(5) Analyses that rely on results of Table 2.1-1 or Figures 2.1-1 or 2.1-2 for potential offsite doses or safe decay times should verify that the assumptions of Table 2.1-2 are conservative for the facility under review. X/Q values should be derived from analysis of on-site meteorological measurements based on 5% worst meteorological conditions	N/A
(6) Analyses should be based on an elastic-plastic curve that represents a true stress-strain relationship	If the analyses is based on an elastic-plastic curve, it must represent a true stress-strain relationship.

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Comparison of Industry Initiative with NUREG-0612 Appendix A

Appendix A 1. General Considerations (cont'd)	Initiative Analysis (cont'd)
(7) The analysis should postulate the "maximum damage" that could result, i.e., the analysis should consider that all energy is absorbed by the structure and/or equipment that is impacted	The analysis will consider the "maximum damage" caused by the transfer of energy to the vessel and supports. Analysis that accounts for appropriate consideration of conservation of momentum is acceptable. It is also acceptable to consider damping.
(8) Loads need not be analyzed if their load paths and consequences are scoped by the analysis of some other load	N/A
(9) To overcome water leakage due to damage from a load drop, credit may be taken for borated water makeup of adequate concentration that is required to be available by the technical specifications	To overcome water leakage due to damage from a load drop, credit may be taken for borated water makeup of adequate concentration that is required to be available by the technical specifications
(10) Credit may not be taken for equipment to operate that may mitigate the effects of the load drop if the equipment is not required to be operable by the technical specifications when the load could be dropped	N/A

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Comparison of Industry Initiative with NUREG-0612 Appendix A

Appendix A 2. Rx Vessel Head Drop Analysis	Initiative Analysis
*These guidelines only consider the dropping of the RV head assembly during refueling and do not apply directly to dropping of the reactor internals such as the steam dryer (BWR), moisture- separator (BWR) or the upper core internals (PWR); however, similar assumptions and considerations would apply to analyses of dropping of reactor internals.	Only Reactor vessel head drop is considered.
(1) Impact loads should include the weight of the reactor vessel (RV) head assembly (including all appurtances), the crane load block, and other lifting apparatus (i.e., the strongback for a BWR).	The analysis should include the weight of the reactor vessel (RV) head assembly below the hook.
(2) All potential accident cases during the refueling operation-. Areas of consideration as a minimum should be:(a) Fall of the RV head from it's maximum height while still on the guide studs followed by impact with the RV flange;(b) Fall of the RV head from its maximum height considering possible objects of impact such as the guide studs, the RV flange, the steam dryer (BWR) or structures beneath the path of travel; and(c) Impact with the fueling cavity wall due to load swing with the subsequent drop of the RV head due to lifting device or wire rope failure.	Area of consideration: Fall of the Reactor vessel head from its maximum height allowed by plant procedures directly (concentrically and flat) on the vessel flange.

Comparison of Industry Initiative with NUREG-0612 Appendix A

Appendix A 2. Rx Vessel Head Drop Anal (continued)	**Initiative Analysis**
(3) All cases which are to be considered should be analyzed in the actual medium present during the postulated accident, e.g., for a PWR prior to reassembly of the reactor, the fueling cavity is drained after the head engages the guide studs to allow for visual inspection of the reactor core control drive rods insertion into the head. During this phase it should be considered that the head will only fall through air, without any drag forces produced by a water environment.	The analysis will consider the actual medium controlled by plant procedures.
(4) In those Nuclear Steam Supply Systems where portions of the reactor internals extend above the RV flange, the internals should be analyzed for buckling and resultant adverse effects due to the impact loading of the RV head. It should be demonstrated that the energy absorption characteristics (causing buckling failure) of these internals should be such that resultant damage to the core assembly does not cause a condition beyond the acceptance criteria for this analysis.	N/A
(5) Reactor vessel supports should be evaluated for the effects of the transmitted impact loads of the RV head. In the case of PWRs where the RV is supported at its nozzles, the effects of bending; shear and circumferential stresses on the nozzles should be examined. For BWRs the effects of these impact loads on the RV support skirt should be examined.	All components and structures in the load path for the reactor vessel head drop load path will be evaluated to assure deformation is limited, that the core remains covered and that cooling of the core is maintained.
(6) The RV head assembly should be considered rigid and not experience deformation during impact with other components or structures.	The RV head assembly should be considered rigid, but the deformation of components attached to the RV head may be realistically considered.

Comparison of Industry Initiative with NUREG-0612 Appendix A

Appendix A.4 Criticality Considerations	**Initiative Analysis**
4.1 Spent Fuel Pool Neutronics Analysis 4.2 Reactor Core Neutronics Analysis	N/A

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Structural Modeling Techniques

- NEI guidelines will not endorse a specific structural modeling technique
- Licensees will develop the structural model, possibly from one the following approaches:
  – Classical methods (manual analysis)
  – Comparative analysis (comparison with previous analyses)
  – Finite Element Methods

Analysis Considerations

- Elastic – Plastic methods are acceptable
- Energy Balance approaches are acceptable
- Conservation of Momentum evaluation is acceptable to establish initial velocity of the vessel after impact
  - Coefficient of Restitution may be used based on conservative assumptions of mass impacted
  - The impacted mass is limited to the mass of the vessel above the supports
- Additional damping may be included in the analyses to account for losses due to friction or non-linear material behavior not explicitly modeled. If used, 5% critical damping for steel and 7% critical damping for concrete structures is acceptable

Analysis Considerations (cont'd)

- CMTRs/concrete cylinder tests may be used to establish material strength/strain limits
- Aging effects to modify concrete strength are acceptable
- Appropriate dynamic increase factors are acceptable
- Post-buckling behavior is addressed, as applicable

RPV Head Drop Structural Success Criteria

- NUREG-0612 provides the basis for Heavy Load Drop Criteria but does not provide or suggest Structural Success Criteria

- Industry initiative will select criteria for generic application which:
  - Realistically consider the technical issues associated with the head drop analysis
  - Are acceptable in other similar applications for non design basis analyses

Proposed Structural Criteria

- Steel Components
  - RPV, associated nozzles and attached piping required for cooling:
    - ASME Section III, Division 1, Appendix F (latest revision approved by the NRC)
    - ANSI/ANS-58.2-1988
    - ANSI/AISC N690-1994 [Q1.5.8] including supplement 2 (2004)
    - Strain-based criteria when large displacement analysis (post-yield buckling) is considered:
      - Average (through the thickness) equivalent plastic strain $\leq$ 2/3 ε_{usl}
      - Localized (through the thickness at discontinuities) equivalent plastic strain $\leq$ 1.0 ε_{usl}
      - Average or localized (through the thickness) plus peak (surface) equivalent plastic strain $\leq$ 1.0 ε_{usl}

      Where ε_{usl} is the strain at maximum stress

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Proposed Structural Criteria (cont'd)

- Steel Components (cont'd)
  - Supports (and supporting structures)
    - ASME Section III, Division 1, Appendix F (latest revision approved by the NRC)
    - ANSI/ANS-58.2-1988
    - ANSI/AISC N690-1994 [Q1.5.8] including supplement 2 (2004)
    - Strain-based criteria when large displacement analysis is considered:
      - Maximum strain shall not exceed 90% of rupture strain
- Concrete
  - ACI-349-1997, Appendix C (supplemented via RG 1.142 Regulatory Positions 10 and 11 as they apply to impact loading)

Displacement Limited Supporting Structures Failures

- A support or concrete can fail one of the acceptance criteria, but the failure must be shown to be displacement limited, and the coolant loop must be shown to remain leak tight for the imposed additional displacement. (The loop should be shown to meet its acceptance criteria including the added displacement from the displacement limited failure.)

Criteria for Analysis of RC and other Attached Branch Lines

- It may be necessary to evaluate the reactor coolant (RC) loop piping and critical attached branch lines to keep the core covered and cooled.
- Drop analyses performed to date have determined that the RC loop piping and attached branch lines are not controlling for this event. To avoid unnecessary analysis, the following criteria are proposed which can be used in a screening process. Systems which meet these criteria are acceptable without stress analysis.
  - RC loop Parameters
    - The RC loop piping is constructed of austenitic stainless steel
    - The length of the piping from the nozzle safe-end to the first vertical constraint or equipment nozzle safe-end is at least 9 feet for the inlet pipe and 13 feet for the outlet pipe. The nominal inside diameter shall be less than 30 inches
    - The maximum net vertical displacement from the drop accident is less than 2.0 inches

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Criteria for Analysis of RC and other Attached Branch Lines (cont'd)

- RC branch line parameters
  - The branch line is constructed of austenitic stainless steel
  - The maximum net vertical displacement at the branch nozzle is less than 1.5 inches
  - There are no active vertical constraints, or interferences, on the branch piping within 5 pipe diameters, or 5 feet, whichever is longer, of the branch nozzle to pipe safe-end weld.
- Criteria, as presented here, are based on typical Westinghouse design NSSS system. It may also be applicable to other systems which meet this criteria.

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In Conclusion, the Initiative Load Drop Analysis Will:

- Demonstrate that after the reactor vessel head drop, the core remains covered with coolant and sufficient cooling is available
- Analyze the RPV head drop in its worst case, the concentric drop directly on the flange from the highest point through the medium allowed by station procedures
- Apply reasonable analysis methods and acceptance criteria
- Apply realistic assumptions for this non design basis event
- Employ comparative analysis, classical approaches or FEM

Next Steps

- Obtain NRC feedback
- Create NEI Guideline document
- Obtain NRC agreement
- If necessary, plants perform load drop analyses and retain on site
- (Complete remainder of NEI Guideline, obtain NRC agreement and complete industry initiative)

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